April 2, 2007

Mr. Christopher M. Crane
President and CEO
AmerGen Energy Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: THREE MILE ISLAND STATION, UNIT 1 - NRC COMPONENT DESIGN BASIS INSPECTION REPORT 05000289/2007006

Dear Mr. Crane:

On February 16, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Three Mile Island, Unit 1 (TMI) facility. The enclosed inspection report documents the results of the inspection, which were discussed on February 16, 2007, with Mr. Rusty West, Site Vice President, and other members of your staff, and during a subsequent telephone conversation on March 15, 2007, with Mr. Craig Smith, Manager of Regulatory Assurance.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. In conducting the inspection, the team examined the adequacy of selected components and operator actions to mitigate postulated transients, initiating events, and design basis accidents. The inspection also reviewed AmerGen's response to selected operating experience issues. The inspection involved field walkdowns, examination of selected procedures, calculations and records, and interviews with station personnel.

This report documents two NRC-identified findings which were of very low safety significance (Green). The findings were determined to involve violations of NRC requirements. However, because of the very low safety significance of the findings and because they were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any of the NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspectors at Three Mile Island.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

Docket No. 50-289
License No. DPR-50

Enclosure: Inspection Report 05000289/2007006

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REGION I

Docket Nos.:	50-289
License Nos.:	DPR-50
Report Nos.	05000289/2007006
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Three Mile Island Station, Unit 1
Location:	PO Box 480 Middletown, PA 17057
Dates:	January 8 to February 16, 2007
Inspectors:	F. Arner, Senior Reactor Inspector (Team Leader) J. Kulp, Reactor Inspector K. Mangan, Senior Reactor Inspector G. Ottenberg, Reactor Inspector (Trainee) A. Patel, Reactor Inspector L. Hajos, NRC Electrical Contractor S. Spiegelman, NRC Mechanical Contractor
Approved by:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000289/2007006; 01/08/2007 - 02/16/2007; AmerGen Energy Company, LLC; Three Mile Island, Unit 1; Component Design Bases Inspection.

The report covers the Component Design Basis Inspection conducted by a team of five NRC inspectors and two NRC contractors. Two findings of very low risk significance (Green) were identified, and considered to be non-cited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified Findings.

Cornerstone: Mitigating Systems

- Green. The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. Specifically, the licensee could not provide demonstrative evidence that design control measures had been established to verify the capability of an operator to manually operate the emergency feedwater control valves. This operator action is a credited licensing basis safety function implemented for certain postulated events involving the loss-of-instrument air and the depletion of the backup air bottle system. The action to take local manual control of the valves to maintain steam generator water level following a loss-of-instrument air is procedurally driven by the Emergency Operating Procedures, described in the Updated Final Safety Analysis Report and credited in the plant Probabilistic Risk Assessment. AmerGen entered this issue into the corrective action program for resolution. Following the completion of the onsite inspection, AmerGen successfully performed a test that demonstrated the capability of an operator to cycle the valves manually.

  The finding is more than minor because it is associated with the design control attribute of the Mitigating System Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was determined to be of very low safety significance (Green) since it did not result in a loss of system safety function. (Section 1R21-.2.1.9)

- Green. The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10CFR50, Appendix B, Criterion III, Design Control. Specifically, AmerGen did not consider the effects of frequency variation on diesel generator loading. The EDG loading calculations of record failed to account for increased loading that would result from allowable frequency variations of up to 61 Hertz (Hz). The existing allowable EDG frequency range within operating and surveillance procedures had not been accounted for in the EDG loading analyses of record. AmerGen entered this issue into the corrective action program and plans to develop a diesel loading calculation revision, in addition to evaluating the existing frequency setting and testing range to ensure consistency between the analyses and procedures.

This design control weakness was considered to be more than minor because it is associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure via analyses the capability of systems that respond to initiating events to prevent undesirable consequences. The finding was determined to be of very low safety significance (Green), since it did not result in a loss of system safety function. The issue had a crosscutting performance aspect in the area of Problem Identification and Resolution. Specifically, a previous concern relative to EDG frequency tolerance in the low direction had been identified and evaluated in the corrective action program (Reference IR 551313), without identification or consideration for any impact due to allowable frequency tolerance and nominal setpoints above 60 Hz on EDG loading. (Section 1R21-.2.1.12)

B. <u>Licensee-identified Violations</u>.

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (IP 71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and operator actions for review using information contained in the Three Mile Island (TMI) Probabilistic Risk Assessment (PRA) and the U.S. Nuclear Regulatory Commission's (NRC) Standardized Plant Analysis Risk (SPAR) models. Additionally, the TMI Significance Determination Process (SDP) Phase 2 Notebooks, Revision 2, were referenced in the selection of potential components and actions for review. In general, the selection process focused on components and operator actions that had a risk achievement worth (RAW) factor greater than 2.0 or a Risk Reduction Worth (RRW) factor greater than 1.005. The components selected were located within both safety-related and non-safety related systems, and included a variety of components such as turbines, pumps, valves, generators, transformers and batteries. There were 11 mechanical and 6 electrical components selected for review.

The team initially compiled a list of a nominal 50 components and 10 operator actions based on the risk factors previously mentioned. The team performed a margin assessment to narrow the focus of the inspection to 17 components and 5 operator actions. The team's evaluation of possible low design margin included consideration of original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition/equipment reliability issues. The assessment included items such as failed performance test results, corrective action history, repeated maintenance, maintenance rule (a)1 status, operability reviews for degraded conditions, NRC resident inspector input of equipment problems, plant personnel input of equipment issues, and industry operating experience. Consideration was also given to the uniqueness and complexity of the design and the available defense-in-depth margins. The margin review of operator actions included complexity of the action, time to complete action and extent of training on the action.

This inspection effort included walk-downs of selected components, including a review of selected simulator scenarios. It also included interviews with operators, system engineers and design engineers, and reviews of associated design documents and calculations to assess the adequacy of the components to meet both design bases and risk informed beyond design basis functions. A summary of the reviews performed for each component, operator action, operating experience sample, and the specific inspection findings identified are discussed in the following sections of the report. Documents reviewed for this inspection are listed in the attachment.

.2 Results of Detailed Reviews

.2.1 Detailed Component Design Reviews (17 Samples)

.2.1.1 Low Pressure Injection (LPI) Pump, DH-P-1A

a. Inspection Scope

The low pressure injection pump was reviewed to ensure that adequate pressure and flow could be developed for transient and accident conditions consistent with the design bases and regulatory requirements. This included review of the net positive suction head (NPSH) analysis from both the borated water storage tank (BWST) and reactor building sump suction paths. The team reviewed past operating experience as reflected in action requests and issue reports, and interviewed AmerGen engineers to discuss past pump performance, pump modifications, air/voiding concerns and associated corrective actions. Additionally, the lubrication and cooling systems for both the pump and motor were reviewed to ensure adequate operation under all conditions.

b. Findings

No findings of significance were identified.

.2.1.2 High Pressure Injection (HPI) Pumps, MU-P-1A/B

a. Inspection Scope

The high pressure injection/makeup pumps were reviewed to ensure that adequate pressure and flow could be developed for normal makeup operation including reactor coolant pump seal injection and flow to the cold leg for accident conditions, consistent with design bases and Updated Final Safety Analysis Report (UFSAR) commitments. This included review of the NPSH analysis and the system flowrate analysis for various system flowpath alignments. The team reviewed issue reports and action requests to review the history of pump performance. The responsible engineers were interviewed to review the assumptions for pump performance utilized in the current transient and accident analyses, and validate that the minimum pump performance expected in those cases was met through testing criteria. The review included ensuring calculations of record were consistent with inservice testing (IST) requirements and operational procedures.

b. Findings

No findings of significance were identified.

.2.1.3 Reactor Building (RB) Sump Suction Header Isolation Valve, DH-V-6A

a. Inspection Scope

The team selected the ‘A’ reactor building sump to LPI pump suction header isolation valve as a representative sample of the TMI sump isolation valves. The team reviewed design documents, including drawings, calculations, procedures, tests and modifications to evaluate the functional requirements of the valve. The team reviewed these documents to ensure the valve was capable of meeting design bases requirements, with consideration of degraded voltage and maximum postulated temperature conditions. The team interviewed design engineers and the motor operated valve (MOV) program manager, and reviewed system health and related issue reports to ensure the valve was capable of performing its design function. The team performed walkdowns of the decay heat (DH) vault area in the auxiliary building, where the valve is located, to assess the consistency between field conditions and assumptions in analyses. Surveillance test results were reviewed to determine whether margin was sufficient to ensure design bases assumptions could be achieved.

b. Findings

No findings of significance were identified.

.2.1.4 High Pressure Injection/Makeup (MU) Pump Recirculation Isolation Valve, MU-V-36

a. Inspection Scope

The team reviewed design and engineering documentation, including calculations and evaluations and selective testing records related to the recirculation isolation valve. The review included calculations of required system pressures under which the valve must perform its safety function, required valve performance such as thrust capabilities, and valve test results. The team walked down accessible portions of the makeup system to assess material condition and system alignment. Additionally, the team reviewed system health reports, system drawings, and MU system corrective action documents to ensure there were no outstanding issues which would affect the design function of the valve.

b. Findings

No findings of significance were identified.

.2.1.5 Decay Heat Removal Cooler, DH-C-1A

a. Inspection Scope

The decay heat system cooler (heat exchanger), DH-C-1A, was reviewed to ensure that the component was capable of meeting its design requirement performance capability and was operated consistent with AmerGen and vendor procedures. Design bases document information, the UFSAR, the Technical Specifications, test results, calculations and analyses were reviewed to ensure consistency with the design bases and regulatory requirements. The team reviewed historical issues relative to indications of boron leakage and hydraulic vibration, and data associated with cleanliness and heat removal capability. This review included verification that tube plugging criteria was consistent with minimum expected heat exchanger performance and the ability to remove design bases heat loads.

b. Findings

No findings of significance were identified.

.2.1.6 Power Operated Relief Valve (PORV), RC-RV-2

a. Inspection Scope

The team reviewed the PORV to ensure consistency and compliance with the UFSAR and Technical Specifications, respectively. The system engineer was interviewed to discuss past operability issues, changes to test procedures and periodic inspection results. The testing was reviewed with respect to evaluating whether the PORV would remain functional for postulated transient and accident conditions, including minimum voltage supply conditions.

b. Findings

No findings of significance were identified.

.2.1.7 Turbine Driven Emergency Feedwater (TDEFW) Pump, EF-P-1

a. Inspection Scope

The team inspected the turbine driven emergency feedwater (TDEFW) pump to ensure the adequacy of its design and ability to perform as required during design basis accident conditions. The team reviewed piping and instrument diagrams (P&IDs), design calculations, system operating procedures, test procedures, summaries of test results and issue reports related to the TDEFW pump design and operation. The team also reviewed the capability of support systems and equipment to provide adequate heat removal and motive force to allow the pump to operate properly during transient or accident conditions. The team also performed a walkdown of the pump and associated equipment to assess material condition and system alignment. The team verified the capability of the pump to provide the required flow during transient and accident conditions.

b. Findings

No findings of significance were identified.

.2.1.8 Motor Driven EFW Pump, EF-P-2B

a. Inspection Scope

The team inspected the 2B Motor Driven Emergency Feedwater Pump to ensure the adequacy of its design and ability to perform as required during design basis accident conditions. The team reviewed piping and instrument diagrams (P&IDs), design calculations, system operating procedures, test procedures, summaries of test results and issue reports related to the pump design and operation. The team also reviewed the capability of support systems and equipment to provide adequate heat removal and motive force to allow the pump to operate properly during accident conditions. The team performed a walkdown of the pump and associated equipment to assess material condition and system alignment. The team verified the capability of the pump to provide the required flow under transient and accident conditions.

b. Findings

No findings of significance were identified.

.2.1.9 Emergency Feedwater Flow Control Valve, EF-30B

a. Inspection Scope

Valve EF-V30B was selected as a representative sample of the EFW flow control valves. The team inspected the valve to verify that it was capable of meeting its design basis requirements. The valve has an active safety function in the closed position. It also had a risk significant function in the open position to support the transient heat removal function. The team reviewed the capability of the valve to be operated in both directions to support transient and accident scenarios. The team reviewed P&IDs, design calculations, system operating procedures, inservice test procedures, summaries of test results and issue reports. The team interviewed operators and the system engineer to gain an understanding of the overall reliability and operation of the valve.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. Specifically, the licensee could not provide demonstrative evidence that design control measures had been established to verify the capability of an operator to manually operate the emergency feedwater control valves. This operator action is a credited licensing basis function implemented for certain postulated events involving the loss-of-instrument air and the depletion of the backup air bottle system.

Description: The team identified that the licensee did not have analysis or a testing program to demonstrate the capability of an operator to take local manual control of air-operated emergency feedwater (EFW) flow control valves, EF-V-30A/B/C/D, following a loss-of-instrument air event. The action to take local manual control of these valves to maintain steam generator water level is procedurally driven by the Emergency Operating Procedures, described in the Updated Final Safety Analyses Report and credited in the Probabilisitic Risk Assessment documents. The EF-V30 valves were designed to fail shut on loss-of-instrument air to prevent an uncontrolled, excessive cooldown of the reactor coolant system. In addition to the normal air supply, the valves have a back-up bottled air supply system designed to provide two hours of air for valve operation. However, after the backup air supply is depleted, local manual operator action is necessary to continue plant cooldown. This would require an operator to manually open and throttle the valves to allow injection of emergency feedwater into the steam generators and to control steam generator level.

In response to questions from the team, AmerGen asked the valve vendor to determine the force required on the handwheel operator to position the valve against the spring force. Using vendor supplied information for a new valve, AmerGen performed a preliminary, informal analysis to determine if an operator was capable of generating this amount of force on the handwheel in its installed orientation. The team reviewed the information AmerGen engineers provided and determined that nominally 50% of the targeted population of operators should be able to operate the valve. The team noted that allowances were not made for degradation in the operating characteristics of the manual operator or valve stem.

The team reviewed the analysis and concluded that there was reasonable assurance that the valves could be manually cycled, given other information such as air operated test data showing the capability of the disc and stem to cycle. However, the team questioned the lack of data showing the capability of a single operator to cycle the valve using the attached handwheel. The team noted that procedure OP-TM-EOP-010 - Guide 16.3 governed the manual operation of the EF-V30 valves, but was ambiguous as to the number of personnel that would be required to respond in order to operate them. Additionally, the team was concerned that the vertical location of the handwheel in the currently installed configuration of the valve may challenge the capability of an operator to cycle the valve without tools or additional leverage. The vendor best estimate maximum required torque for the valves was 114 ft-lbs.

AmerGen initiated an issue report in the corrective action program to document the team's concern. Following the completion of the onsite inspection, AmerGen successfully developed and performed a test procedure that demonstrated the capability of an operator to cycle the valve manually. The test was well developed with adequate controls in place and showed that the valve could be operated to the maximum expected position required, 68% open, with moderate difficulty.

Analysis: The performance deficiency associated with this finding was that the licensee did not have analysis or a testing program to demonstrate the capability of an operator to take local manual control of the EFW air-operated flow control valves. This is a credited function for events which result in the air supply being depleted.

The finding is more than minor because it is associated with the design control attribute of the Mitigating System Cornerstone and inadequate design control measures affect the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team assessed this finding in accordance with NRC Manual Chapter 0609, Appendix A, Attachment 1, Significance Determination Process (SDP) for Reactor Inspection Findings for At-Power Situations, and determined that it was of very low safety significance (Green) since it did not result in a loss of any system safety function. Specifically, AmerGen performed a test on March 3, 2007, which demonstrated that a plant operator would be able to successfully cycle the valves using the attached manual handwheel.

Enforcement: 10 CFR 50 Appendix B, Criterion III, Design Control, requires, in part, that design control measures provide for verifying or checking the adequacy of design via simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, prior to March 3, 2007, the adequacy of the design function for manually operating the valves had not been verified or checked through simplified calculational methods or by the performance of a suitable testing program. Because the finding was of very low safety significance and has been entered into AmerGen's corrective action program (IR 591795), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000289/2007006-01, Inadequate Verification of Manual Capability for EFW Flow Control Valves)**

.2.1.10 Station Blackout Diesel Generator Cooling System

a. Inspection Scope

The team inspected the cooling water system for the station blackout (SBO) diesel to ensure the adequacy of its design and ability to perform as required during station blackout conditions. The team reviewed piping and instrument diagrams (P&IDs), design calculations, system operating procedures, and issue reports related to the design and operation of the SBO diesel. The team verified the capability of the system to provide the required cooling water flow to support operation of the SBO diesel under the most limiting conditions such as elevated river water temperatures.

b. Findings

No findings of significance were identified.

.2.1.11 RCP Seal Injection RB Isolation Valve, MU-V-20

a. Inspection Scope

The team reviewed the capability of the air operated MU-V-20 valve which provides containment isolation for the 4-inch supply line for the reactor coolant pump (RCP) seals. The documents which were reviewed included valve and system design calculations, inservice testing reports and data, vendor manual, vendor valve and system process diagram drawings, and associated Issue Reports related to this valve. The team reviewed compensatory actions evaluated by AmerGen with respect to air leakage and air accumulator capacity and the affect on valve capability.

b. Findings

No findings of significance were identified.

.2.1.12'B' Emergency Diesel Generator (EDG), EG-Y-1B

a. Inspection Scope

The team reviewed the electrical capabilities of the 1B Emergency Diesel Generator. The evaluation of 1B EDG focused on its ability to power safety related loads during design basis events. Specifically, the team reviewed: 1) the load flow analysis and voltage drop calculations to verify that adequate voltage was provided to the safety-related loads during worst-case loading conditions; 2) the ability of the EDG to support the maximum loading condition under a loss-of-offsite power and loss-of-coolant accident scenario; and 3) the sequential starting of loads to determine if 1B EDG had sufficient capability to accelerate the loads within the time periods specified in the calculation of record. The team reviewed the EDG test results to verify that the test conditions verified compliance with technical specification requirements. The team conducted a detailed walkdown of the 1B EDG. The team reviewed design documents, calculations, inservice test (IST) criteria and results, surveillance testing, vendor manuals, maintenance history, licensing basis documents, and associated Issue Reports.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, in that AmerGen did not consider the effects of frequency variation on emergency diesel generator loading. Specifically, the EDG loading calculations of record did not account for increased loading that would result from allowable frequency variations of up to 61 Hertz (Hz).

Description: The EDG loading calculation, C-1101-741-E510-005, Revision 4, Loading Summary of Emergency Diesel Generators & Engineered Safeguards Buses, determined diesel loading based on maximum loads during the design large break loss-of-coolant accident (LOCA). The loading was based on nominal 60Hz operation of pumps and fans, and did not account for the frequency variation allowed by operating procedure, OP-TM-861-902, Diesel Generator EG-Y-1B Emergency Operations. This procedure allows operation up to 61Hz. The team noted that mechanical affinity laws show that power demanded by centrifugal pumps and fans increases by the cube of the ratio of the speeds. Since the diesel loading during an accident is comprised primarily of centrifugal loads, the team determined this phenomenon should have been considered in the bounding loading calculations. Additionally, the team noted that technical specification surveillance procedure, 1303-4.16, Emergency Power System, has an acceptable range from 60.2 to 61 Hz with a midrange value of 60.6 Hz.

AmerGen entered this issue into their corrective action program and reviewed the potential impact of the higher frequency on loading of the machine and on fuel oil storage requirements. Issue Report 581933 was a preliminary assessment which determined that the increased frequency allowed by the procedure would increase the design basis loading from a nominal 2758 kW to 2898 kW. The increased loading remained within the 2000 hour 3000 kW rating of the machine and the assessment concluded there was not an adverse impact on operability.

During the review of the EDG loading capability, the team noted several other weaknesses which indicated a lack of consistency within calculations of record, UFSAR tables, and test results:

- UFSAR Tables 8.2-8 and 8.2-9a which showed major EDG loads, were inconsistent with respect to the diesel loading calculation of record (C-1101-741-E510-005). Issue Report 577954 was initiated during the inspection to address this issue.

- The team identified that an issue was in the corrective action system from November of 2005, where testing had been performed on an EFW pump motor which indicated an actual load of 447 kW. This was in contrast to the EDG loading calculation of record which indicated 406kW. This information had not been resolved with respect to the impact on EDG loading calculations of record prior to this inspection. (IR 590832)

The team reviewed AmerGen's operability assessment (IR 581933) and determined there was no operability impact due to the higher allowable frequency considerations. This review included AmerGen's evaluation (IR 590832) of the impact from the above stated EFW pump motor issue, relative to EDG loading considerations. AmerGen's preliminary corrective action plan was to develop a diesel loading calculation revision along with evaluating the existing frequency setting and testing range to ensure consistency between the analyses and procedures. This issue was also applicable to EDG EG-Y-1A.

Analysis: The performance deficiency associated with the issue is that the allowable EDG frequency range permitted in operating and surveillance procedures had not been accounted for in the EDG loading analyses of record. This design control weakness was considered to be more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone. Inadequate design control measures affect the cornerstone objective to ensure the capability of systems that respond to initiating events to prevent undesirable consequences. The team assessed this finding in accordance with NRC Manual Chapter 0609, Appendix A, Attachment 1, Significance Determination Process (SDP) for Reactor Inspection Findings for At-Power Situations, and determined that it was of very low safety significance (Green) since it did not result in a loss of any system safety function.

The issue had a crosscutting performance issue in the area of problem identification and resolution. Specifically, a previous concern relative to EDG frequency tolerance in the low direction had been identified and evaluated in the corrective action program (IR 551313), without identification or consideration for any impact due to allowable frequency tolerance and nominal setpoints above 60 Hz on EDG loading.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, Design Control, requires that measures shall provide for verifying or checking the adequacy of design. Contrary to this requirement, prior to January 22, 2007, AmerGen's calculations of record did not adequately verify or check the adequacy of EDG loading within the procedurally allowable frequency limits. Specifically, EDG loading calculation of record, C-1101-741-E510-005, Revision 4, did not account for allowable variations in EDG frequency. Operating procedure, OP-TM-861-902, and Surveillance Procedure, 1303-4.16, allowed EDG frequency up to 61 Hz. Because this violation was of very low safety significance and was entered into AmerGen's corrective action program as IR 581933, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000289/2007006-02, Inadequate Design Control for EDG Loading)**

.2.1.13 4160 Volt Alternating Current (Vac) Safety Bus, 1E

a. Inspection Scope

The team reviewed calculations and drawings to determine if the loading of 4160V vital bus 1E was within equipment ratings. The team reviewed the adequacy and appropriateness of design assumptions and calculations related to motor starting and loading voltages to determine if the voltages across motor terminals, under worst case motor starting and loading conditions, would remain above the minimum acceptable values. On a sample basis, the team reviewed maintenance and test procedures and acceptance criteria to verify that 4160V Vital Bus 1E was capable of supplying the minimum voltage necessary to ensure proper operation of connected equipment during normal and accident conditions. The team reviewed the adequacy of the short circuit ratings of the switchgear and circuit breakers, and the adequacy of protective device coordination provided for a selected sample of equipment.

The team reviewed calculations, drawings, and procedures to determine whether undervoltage relay setpoints, load shed schemes, and load sequencing, were adequate to ensure availability of vital loads within the times assumed in Section 8.2.2 of the UFSAR. The team conducted a walkdown of 4160V vital buses to determine if their material condition and operating environment were consistent with the design basis, and to verify that system alignments were consistent with the design basis. The team reviewed the licensing commitments contained in the UFSAR to determine the requirements for the settings of the degraded and loss of voltage relays. The team reviewed the basis for the setpoint, whether the settings allowed for proper operation of all loads, and whether all required loads were analyzed to operate successfully during the period of set time delay.

b. Findings

No findings of significance were identified.

.2.1.14 480 Volt Alternating Current (Vac) Engineered Safeguards Bus, 1P

a. Inspection Scope

The team reviewed calculations and drawings to determine if the loading of 480V vital bus 1P was within equipment ratings. The team reviewed the adequacy and appropriateness of design assumptions and calculations related to motor starting and loading voltages to determine if the voltages across motor terminals, under worst case motor starting and loading conditions, would remain above the minimum acceptable values. On a sample basis, the team reviewed maintenance and test procedures and acceptance criteria to verify that 480V vital bus 1P was capable of supplying the minimum voltage necessary to ensure proper operation of connected equipment during normal and accident conditions. The team reviewed the adequacy of the short circuit ratings of the switchgear and circuit breakers, and the adequacy of protective device coordination provided for a selected sample of equipment.

The team reviewed calculations, drawings, and procedures to determine whether undervoltage relay setpoints, load shed schemes, and load sequencing, were adequate to ensure availabiity of vital loads within the times assumed in Section 8.2.2 of the UFSAR. The team conducted a walkdown of 480V vital buses to determine if their material condition and operating environment were consistent with the design basis, and to verify that system alignments were consistent with the design basis.

b. Findings

No findings of significance were identified.

.2.1.15 Auxiliary Transformer, 1B

a. Inspection Scope

The team reviewed calculations, drawings, modification packages, maintenance procedures, and vendor data to determine whether the auxiliary transformers that supply power from offsite to engineered safety feature buses were adequately designed and maintained. Specifically, the team reviewed load flow calculations to determine whether loading of buses and transformers was within their ratings. The team also reviewed the design of protective relaying for buses and transformers to determine whether equipment was properly protected, and also not subject to spurious tripping under expected transient and steady state loading conditions. The team reviewed transformer vendor manuals, design change packages and surveillance test results to verify that applicable test acceptance criteria and test frequency requirements were met.

b. Findings

No findings of significance were identified.

.2.1.16 250/125 Vdc Battery, EED-B-1A

a. Inspection Scope

The team reviewed the station battery calculations to verify that the battery sizing would satisfy electrical loading requirements and that the minimum possible voltage was taken into account. The team focused on verifying that the battery and battery chargers were adequately sized to supply the design duty cycle of the 250/125 Vdc system for both the loss-of-offsite power/loss-of-coolant accident and station blackout loading scenarios, and that adequate voltage would remain available for the individual load devices required to operate during a two hour coping duration. The team reviewed battery test results to verify that applicable test acceptance criteria and test frequency requirements specified in Technical Specifications were met. In addition, a walkdown was performed to visually inspect the physical and material condition, and component readiness of the battery and battery chargers.

b. Findings

No findings of significance were identified.

.2.1.17 EDG 'A' Day Tank Level Switch, EG-LS 244A

a. Inspection Scope

The team selected EDG 'A' day tank level switch, EG-LS 244A, as a representative sample of level switches used in both EDG day tanks. The level switch is a three level unit that starts the AC and DC auto transfer pumps which transfer fuel oil from the

30,000 gallon fuel oil storage tank to the day tank. The team reviewed design documents, inservice test (IST) criteria and results, surveillance testing, calibration records, vendor manuals, maintenance history, licensing basis documents, and Issue Reports to verify the switch was appropriately set and tested.

b. Findings

No findings of significance were identified.

.2.2 Review of Low Margin Operator Actions (5 samples)

The team performed a risk assessment of expected operator actions, and selected a sample of operator actions for detailed review based upon potential low margin for successful completion of the action. Low margin issues were generally characterized as having one or more of the following attributes:

- Low margin between the time required and time available to perform the actions;
- Complexity of the actions;
- Reliability or redundancy of the components associated with the actions;
- Procedure or training challenges that may impact the operators' ability to perform the actions; and
- Extent of actions to be performed outside of the control room

.2.2.1 Operator Aligns Station Blackout (SBO) Diesel Generator from Control Room

a. Inspection Scope

The team selected the operator actions to manually align the station blackout diesel generator (DG) to supply emergency AC power to one of the 4160 Vac emergency buses during a SBO event. These operator actions enable the SBO DG to supply all loads required to mitigate the effects of a SBO. The team noted that the SBO analysis requires that the SBO DG energize an emergency bus within ten minutes upon a loss-of offsite power and failure of both emergency diesel generators. In order to evaluate the time required to perform all necessary manual actions, the team interviewed operators and training personnel. The team observed a simulator scenario, in which the operators were required to energize an emergency bus with the SBO diesel generator, to verify that the procedure was adequate to ensure the bus could be energized within ten minutes as assumed by the analysis. Finally, the team evaluated the available time margins to perform the operator actions in order to verify AmergGen's operating and risk model assumptions.

b. Findings

No findings of significance were identified.

.2.2.2 Operator Refills the Borated Water Storage Tank (BWST) following a Steam Generator Tube Rupture (SGTR)

a. Inspection Scope

The team selected the manual actions associated with the alignment of the spent fuel pool recirculation pump to refill the BWST from the spent fuel pool. The team determined that the action is required to ensure sufficient inventory is provided to make up for the loss of primary inventory that is lost to the secondary system. The team reviewed a variety of SGTR scenarios to determine the flowrate to the BWST required to maintain sufficient primary coolant during the event. Additionally, the team walked down the equipment lineup to ensure components were accessible during the event. The Human Reliability Analysis assigned a stress level of high with these actions as the procedure would require an operator to manually operate valves in various locations throughout the auxiliary building. Finally, the team evaluated the available time margins to perform the operator actions to verify operating and design basis assumptions were correct, and reviewed the applicable procedures to assess the guidance provided to the operator.

b. Findings

No findings of significance were identified.

.2.2.3 Operator Transfers Suction from BWST to Containment Sump following LOCA

a. Inspection Scope

The team selected the manual actions to realign low pressure and/or high pressure injection (piggy back mode) from the BWST to the containment sump following a loss-of coolant accident. Failure of these actions would result in loss of suction to the low pressure injection (LPI) pumps and the failure to maintain inventory, to the loss of reactor pressure vessel (RPV) level control and core uncovery. The team reviewed calculations to determine required NPSH and minimum vortexing levels for both the BWST and containment sump. Additionally, the team reviewed the calculations used to determine containment level to assess the adequacy of the calculation and determine what the minimum containment level would be at the swap-over. In order to evaluate the time requirements to perform the manual actions, the team observed a simulator scenario to verify that procedures were sufficient to ensure the actions could be completed such that NPSH was maintained for the LPI pumps.

b. Findings

No findings of significance were identified.

.2.2.4 Operator Actions to Supply Alternate Cooling following a Loss of River Water

a. Inspection Scope

The team selected the manual actions to align an alternate cooling supply to the makeup pumps following a loss of river water. Failure of these actions could result in loss of RPV level control and core uncovery. In order to evaluate the time requirements to perform the manual actions, the team interviewed operators and training personnel. The team performed field walkdowns to independently verify that equipment was staged as required and if followed, the procedure provided sufficient guidance to ensure the appropriate alignment was completed. The team evaluated the available time margins to perform the operator actions to verify operating and risk model assumptions. Finally, the team reviewed AmerGen's evaluation of the staged equipment to ensure that adequate cooling would be provided to the make-up pumps.

b. Findings

No findings of significance were identified.

.2.2.5 Manual Control Emergency Feedwater Throttle Valves following a Loss of Instrument Air

a. Inspection Scope

The team selected the manual actions to manually throttle EFW injection valves following the loss-of-instrument air event. The action requires operators to manually position the normally air operated valves to control steam generator level. The team interviewed licensee personnel to determine if operators had ever operated the valves manually. Additionally, the team requested an evaluation to determined how much force would be required to turn the valve handwheel. The team performed field walkdowns to independently assess the ability of operators to access the valves and determine the task complexity. Finally, the team reviewed the applicable procedures to assess the adequacy of the guidance provided to the operator.

b. Findings

The team identified a concern relative to the capability of an operator to locally manually operate the valves which is documented in section .2.1.9 of this report. No other findings of significance were identified.

.3 Review of Industry Operating Experience (OE) and Generic Issues (5 Samples)

a. Inspection Scope

The team reviewed selected OE issues for applicability at the Three Mile Island Unit 1 facility. The team performed a detailed review of the OE issues listed below to verify that the licensee had appropriately assessed potential applicability to site equipment and implemented corrective actions as required.

NRC Information Notice (IN) 2005-30: Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design

The team reviewed the potential of inadequate engineering design assessment of the effect of internal flooding that could cause the inoperability of multiple safety related components. The areas reviewed included buildings adjoining the turbine building to assess the possible effects that a circulating water pipe failure could have on plant equipment. The team conducted a walkdown of susceptible areas to perform the assessment. Additionally, the team reviewed the assessment performed by AmerGen to determine what actions the site had taken following issuance of the information notice to evaluate the adequacy of the actions taken.

NRC Information Notice 1995-37: Inadequate Offsite Power System Voltages During Design Basis Events

The team reviewed the applicability and disposition of NRC IN 95-37. The basis of IN 95-37 was a concern for circumstances that could result in inadequate offsite power system voltages during design basis events. The failure to periodically update the original voltage analyses as the result of changing offsite grid or plant conditions could result in unintentional operation outside regulatory requirements. The Information Notice had identified the potential that the setpoints of undervoltage protection relays may require changes to ensure adequate voltages at the terminals of all safety related equipment. The team reviewed AmerGen's offsite procedures, design of the undervoltage relays, and preventive maintenance procedures to ensure their adequacy.

NRC Information Notice 1992-40: Inadequate Testing of Emergency Bus Undervoltage Logic Circuitry

The team assessed the licensee's review and disposition of NRC IN 92-40. The team selected this notice to verify the adequacy of the existing test methodology to de-energize the emergency safety buses at the appropriate voltage levels. The team reviewed AmerGen's procedure that verifies the de-energization of the emergency safety buses via the undervoltage relays. Additionally, a walkdown was performed to visually inspect the physical and material condition of the undervoltage relays.

NRC Generic Letter (GL) 1995-07: Pressure Locking and Thermal Binding of Safety Related Power-Operated Gate Valves

The team reviewed the applicability and disposition of GL 95-07. The NRC issued this letter to request that licensee's perform or confirm that they previously performed: (1) evaluations of operational configurations of safety related, power-operated gate valves for susceptibility to pressure locking and thermal binding; and, (2) further analyses and any needed corrective actions to ensure that those valves which were susceptible were capable of performing the safety functions within the current licensing bases of the facility. The team sampled the reactor building sump to LPI pump suction header isolation valves, DH-V-6A/B, and the low pressure injection to reactor coolant system valves, DH-V-4A/B. The team reviewed both the original GL 95-07 evaluation and the current motor operated valve analyses. This review included verifying that the inputs and assumptions used in the original evaluation remained valid.

NRC Bulletin 1988-04: Potential Safety Related Pump Loss

The team reviewed the applicability and disposition of Bulletin 88-04. This bulletin described conditions where potential design deficiencies in the minimum flow lines and interactions between pumps running in parallel in a system, may lead to permanent pump damage due to prolonged operation at or near shutoff head conditions. The team specifically reviewed this bulletin with respect to operation of the emergency feedwater pumps.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

a. Inspection Scope

The team reviewed a sample of problems that were identified by the licensee and entered into the corrective action program. The team reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design or qualification issues. In addition, Issue Reports (IRs) written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4AO6 Meetings, Including Exit

Exit Meeting Summary

On February 16, 2007, the team presented the inspection results to Mr. Rusty West, Three Mile Island, Site Vice President, and other members of the Three Mile Island staff. A subsequent telephone conversation took place on March 15, 2007, between Mr. Lawrence Doerflein, Team Manager, and Mr. Craig Smith, Manager of Regulatory Assurance, to discuss changes since the exit meeting. The team verified that no proprietary information reviewed during the inspection was retained.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

R. West	Site Vice President
T. Dougherty	Plant Manager
P. Bennett	Mechanical Design Engineering Manager
R. Ezzo	Electrical Design Engineering Manager
C. Smith	Regulatory Assurance Manager
B. Masoero	System Engineer
D. Hull	I&C Design Engineer
B. Smith	Electrical Design Engineer
M. Reed	Design Engineering
V. Zeppos	Mechanical Design Engineer

NRC Personnel

W. Cook	Region I Senior Risk Analyst
D. Kern	TMI Senior Resident Inspector
J. Brand	TMI Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000289/2007006-01	NCV	Inadequate Verification of Manual Capability for EFW Flow Control Valves
05000289/2007006-02	NCV	Inadequate Design Control for EDG Loading

LIST OF DOCUMENTS REVIEWED

Calculations

C-1101-101-5321-002, Station Battery Rack Anchorage Design, Rev. 0
C-1101-104-5320-004, Spring Can Calculation for PORV and SRV Discharge Line, Rev. 0
C-1101-202-E270-438, Core and SFP Time to Boil/Uncover, Rev. 1
C-1101-210-E610-011, LPI and BS Pump NPSH Margin Available from the RB Sump, Rev. 5
C-1101-211-5320-004, TMI-1 HPI Flow Analysis, Rev. 1A
C-1101-211-5320-001, TMI-1 HPI System Flow Restriction Analysis, Rev. 1A
C-1101-211-E540-085, TMI Makeup Tank Gas Entrainment Determination, Rev. 0
C-1101-211-E540-091, TMI-1 IST Acceptance Criteria for HPI Pump, Rev. 0
C-1101-211-E610-026, HPI Pump NPSH in piggy-back mode, Rev. 0
C-1101-211-5350-014, MU Pump NPSH - High Suction Temperature, Rev. 0
C-1101-211-5360-005, Cooling of MU Pump by DCCW system when DR system unavailable, Rev. 0
C-1101-211-E410-090, Seismic Adequacy of MU-P-1A/1B/1C Speed Changer Oil piping, Rev. 1
C-1101-212-5360-004, DHR system drop line NPSH, Rev. 1
C-1101-212-5350-005, DHR suction head loss, Rev. 0
C-1101-212-5360-024, Pressure Drop between the BWST and the DHR pump suction, Rev. 0
C-1101-212-E410-70, DH-P-1A & DH-P-1B NPSH minimum water level, Rev. 0
C-1101-212-5360-043, TMI Loads on the ECCS pump during LB LOCA, Rev. 3
C-11-1-212-E540-075, IST Acceptance Criteria for LPI pumps, Rev. 0
C-1101X-212-322C-A64, DHR Pump Remote Vent Piping Analysis, Rev. 0
C-1101-212-E410-081, TMI-1 1R14 DH-C-1A Performance Evaluation, Rev. A
C-1101-212-E410-084, TMI-1 DH-C-1A/B Design Analysis, Rev. 0
C-1101-212-5360-008, Decay Heat Removal System Resistance, Rev. 0
C-1101-212-5310-050, TMI-1 BWST Vortex Determination, Rev. 2
C-1101-210-E610-010, Reactor Building Minimum Level During Recirculation Following a LBLOCA, Rev. 0
C-1101-212-E510-074, Low Pressure Injection (LPI) / Decay Heat (DH) Removal Flow Calibration and Instrument Loop Error, Rev. 1
C-1101-214-E410-019, Building Spray Flow Basis and LPI RB Sump Recirc Mode LPI NPSH / Throttling Basis, Rev. 2
C-1101-220-5360-036, Back Pressure of Pressurizer Safety Valve and PORV, Rev. 0
C-1101-224-E610-069, Station Blackout at 2568 MWt with 20% SG Tube Plugging, Rev. 1
C-1101-224-E610-070, Loss of Feedwater at 2568 MWt with 20% SG Tube Plugging, Rev. 4
C-1101-424-5310-054, EFW Pump Brake Horsepower During LBLOCA/LOOP Event, Rev. 1
C-1101-424-5310-029, TMI-1 Emergency Feed Pump Turbine Performance, Rev. 0
C-1101-424-5350-040, TMI-1 Emerg. FW Flow Loop Error Analysis, Rev. 4
C-1101-424-5360-035, Time Value of EFW Left in CSTs at Lo-Lo-Level Alarm, Rev. 0
C-1101-424-E410-075, TMI-1 EFW Pump NPSH and Suction Design Basis Lineups, Rev. 0
C-1101-424-E420-068, EFW Loop Calibration Tolerances, Rev. 0
C-1101-424-E540-065, TMI-1 IST Acceptance Criteria for EFW Pumps, Rev. 2
C-1101-424-E610-072, CST Heatup following Extended EFW Operation, Rev. 0

C-1101-572-5350-005, RB Sump Minimum Level Setpoints and Baseline Calibration, Rev. 1
C-1101-644-5350-005, TMI-1 Condensate Storage Tank Level Indication Accident Loop Error Analysis, Rev. 1
C-1101-700-E510-010, TMI-1 AC Voltage Regulation Study, Rev. 6
C-1101-700-E510-008, TMI-1 Electrical Impedance Model, Rev. 4
C-1101-720-5350-001, Auxiliary Transformer 87 Relay Current Transformer on Fast Transfer, Rev.1
C-1101-730-5350-001, GL-89-10 MOV's Degraded Grid Voltage Drop Calc., Rev. 9
C-1101-732-5350-005, TMI-1 Protective Relays 4.16 kV Class 1E Switchgear, Rev. 1
C-1101-732-E510-0008, TMI-1 4.16kV ES Bus 1D & 1E Degraded Grid Undervoltage Setpoint Drift Analysis, Rev. 1
C-1101-733-5350-008, Load Reduction on Unit Substations 1P & 1S, Rev.1
C-1101-734-5350-003, TMI-1 Battery Capacity Sizing and Voltage Drop for DC Systems, Rev. 9
C-1101-734-5520-001, Station Battery Hydrogen Generation, Rev. 0
C-1101-734-5320-006, 1A, 1C, & 1E Battery Chargers, Rev. 0
C-1101-741-E420-006, Diesel Generators EG-Y-1A/B Protective Relay Settings, Rev. 1
C-1101-741-E420-007, TMI-1 Emergency Diesel Generator Voltage and Frequency Response, Rev. 0
C-1101-741-5351-003, Relay Settings for Diesel Generator Up-To-Voltage and Thermal Overload Relays, Rev. 0
C-1101-741-E510-005, Loading Summary of Emergency Diesel Generators & Engineered Safeguards Buses, Rev. 4
C-1101-852-5360-004, Two Hour Backup Instrument Air System As-Build Capability Calculation, Rev. 2
C-1101-900-E420-162, ACE AOV Design Basis Capability Calculation for EF-V-30A, Rev. 2
C-1101-900-E410-039, MOV Delta P and Basis, Rev. 8
C-1101-900-E420-161, AOE-AOV Design Basis Capability Calculation for MU-V-V20, Rev. 1
C-1101-900-E220-080, TMI Air Operated Valve Program System Calculations, Rev. 1
1101X-322C-A23, TMI-1 EFW Pump Motor Bearing Oil Temperature Analysis, 3/12/81
TDR 828, TMI-1 Turbine Driven Emergency Feedwater Pump Testing, Rev. 0
V-1101-424-041, Resolution of MS-V22 Lift Problem (EFW), 02/26/87

Surveillance Test Procedures

1301-4.6.1, Station Battery 1A Weekly, 12/28/2006, 1/4/2007 and 1/11/2007
1301-5.8.1, Station Battery 1A Quarterly, 9/14/2006 and 12/14/2006
1301-8.2A, Diesel Generator Inspection (Electrical), 10/4/04
1301-8.2B, Diesel Generator Inspection (Instrumentation), 10/4/04
1302-6, Calibration of Non-Tech Spec Instruments Used for Tech Spec Compliance, 3/31/04
1302-5.30B, EG-Y-1B Diesel Generator Protective Relaying, 10/4/04
1302-5.31A, 4160 V D and E Bus Degraded Grid Undervoltage Relay System Calibration, 12/20/2006 & 9/14/2006
1302-5.31B, 4160 V D and E Bus Degraded Grid Undervoltage Relay System Calibration, 9/3/2004 and 4/11/2006
1302-5.25, Reactor Building Sump Level, Rev. 20, 11/12/05
1303-4.16, Emergency Power System, 11/16/2006

1303-11.11, Station Battery Load Test, 11/7/2003 and 10/30/2005
1303-11.10, ES System Emergency Sequence and Power Transfer Test, 10/24/05
1303-13.4, Remote Shutdown System Functional Test, Rev. 3, 10/26/05, 11/11/05, and 11/15/05
1420-LTQ-7, Dynamic Testing of Motor Operated Valves Using ITI Movats Series 3000 Valve Analysis System, Rev. 8, 03/22/95
1420-LTQ-9, Dynamic Testing of Motor Operated Valves Using Liberty Technology Votes System, Rev. 0, 03/21/95
1420-LTQ-10, MOV Diagnostic Testing with Quicklook, Rev. 6, 10/26/01
OP-TM-211-231, IST of ECCS Bypass Valves - MU Discharge, Rev. 1, 10/11/03
OP-TM-211-231, IST of ECCS Bypass Valves - MU Discharge, Rev. 2, 10/25/05
OP-TM-212-217, DH-V-6A to RB Sump Leak Check and VT-2, IC-14929, 10/28/03
OP-TM-212-217, DH-V-6A to RB Sump Leak Check and VT-2, Rev 3, 11/14/05

Completed Work Orders

R1801878
R2026220
R2097152
R2095697
R2094121

Issue Reports

00108934	00253197	00398749	00577512*	00587670*
00138874	00274510	00398946	00577700	00587630*
00155844	00291825	00425749	00577951*	00588775*
00164725	00292201	00426656	00577954*	00589148*
00165028	00292503	00449885	00579419*	00589155*
00165822	00294896	00449888	00579438*	00589260*
00166682	00297454	00474653	00581933*	00589320*
00167099	00298626	00516156	00582111*	00589480*
00169958	00325067	00518847	00582115*	00589752*
00173331	00325077	00519114	00582611*	00589815*
00178657	00336684	00531174	00582617*	00590774*
00181758	00336276	00535841	00582857*	00590427*
00181985	00352246	00551313	00582909*	00590430*
00186223	00356200	00557924	00585317*	00590466*
00198109	00366490	00562611	00586211*	00590774*
00202762	00395928	00562805	00587449*	00590832*
00212967	00396859	00567178	00587676*	

* Issue Report written as a result of inspection effort

Design Basis Documents

Configuration Change T1-CCD-113202-001, DHV4 Pressure Locking Modification, Rev. 0
Design Change Notice DC0042, Remove Recirc. Orifice from Decay Heat Pumps DH-P-1A & DH-P-1B & Replace w/ Flex. Gasket, Rev. 0
SDBD-TI-211, System Design Basis Document for Makeup and Purification (#211), Rev. 4
SDBD-TI-212, System Design Basis Document for Decay Heat Removal System (#212), Rev. 4

Drawings

11865841, Electrical Schematic AC Auxiliary & Generator, Rev. 25
11865808, Fuel Oil System Schematic, Rev. 5
1E-155-02-002, General Arrangement Control Room Tower Plan Floor Elevation 322 Ft., Rev. 11
206051, Electrical One Line Diagram 250/125V DC System & 120V AC Vital Instrument, Rev. 30
208164, Electrical 4160V Switchgear (1E3), Rev. 25
01760240, Diesel Generator 1A 3-Line & Schematic Diagram, Rev. 25
302-351, Emergency Diesel Generator Services, Rev. 18
DWG 302-082, Emergency Feedwater, Rev. 23
DWG 302-202, Nuclear Services River Water System, Rev. 69
DWG 302-357, Station Blackout Diesel, Rev. 4
DWG 302-610, Nuclear Services Closed Cycle Cooling Water, Rev. 75
DWG 302-640, Decay Heat Removal, Rev. 80
DWG 302-641, Decay Heat Removal – Decay Heat Pumps 1A/B Aux. Systems Flow Diagram, Rev. 6
DWG 302-660, Makeup and Purification, Rev. 42
DWG 302-661, Make-Up and Purification, Rev. 57
E-206022, Electrical One Line & Relay Diagram 4160V ENGD. Safeguards Switchgear, Rev. 21
E-206032, Electrical One Line & Relay Diagram – ENGD. Safeguards Screen House, Reactor Building, H&V, 480V Switchgear, Rev. 15
E-206011, Electrical Main One Line & Relay Diagram, Rev. 49
E-206032, One Line & Relay Diagram - ENGD SFGDS Rev. 15
E-206031, One Line & Relay Diagram - Turb. & Serv. H&V, Rev. 23
E-208164, Electrical 4160V Switchgear (1E3) G11-02 Diesel Generator 1B Breaker, Rev. 25
C-207012, Three Line Diagram Aux. Transformer 1A & 1B Metering and Relaying, Rev. 4
1E-151-02, General Arrangement Turbine Bldg Elev. 305', Rev. 2
1E-710-11, Electrical Schematic One Line Diagram, Rev. 1
208-203 Electrical Elementary Diagram, 4160V Switchgear (ES) (1D3) Motor Driven Emergency FW Pump EF-P2A, Rev. 15
11866062, Fuel Oil Day Tank 550 Gal Nominal, dated 4/14/70
E-211-823, Roof, Floor, and Equipment Drains Diesel Generating Bldg, Rev. 2
E-304-642, Decay Heat Removal Sections and Details, Rev. 31
E-744-004, Site Improvements Plot Plan: Catch Basin Locations Around Plant Buildings, Rev. 20
641-046, Condensate Storage Tank, Rev. 0
K-2813, Diesel Fuel Storage Tank 30,000 Gallons, Rev. 3

D-521-007, Reactor Building- Steel, Sump Liner Detail, Rev. 3
E-311-813, Floor and Equipment Drains, Reactor Bldg. El. 279'0" & El. 262'7", Rev. 18
E-47035, 14" N-2216-SP 300lb Gate Valve, 04/28/70
ID-212-22-1001, Decay Heat Removal DH-V4A/B Pressure Locking Mod., Rev. 1
ID-ISI-FD-005, ISI Boundary Sketch Decay Heat Removal System, Rev.19
ID-ISI-FD-017, ISI Boundary Sketch Makeup and Purification System (Makeup Portion), Rev.18
LS-29659, Level Switch (Type LS-800) (RB Sump), Rev. C
SS-209-097, Electrical Elementary Diagram, DC & Miscellaneous, Rev. 8

Maintenance Procedures

1302-5.31A, 4160V D and E Degraded Grid Undervoltage Relay System Calibration
1302-5.31B, 4160V D and E Loss of Voltage Relay System
1302-5.31D, 4160V D and E Loss of voltage/Degraded Timing Relay Calibration & Logic Check
1450-001, 4160 Volt Motor Feeder Breaker Relay Functional Test

Miscellaneous

03-00304, Replace Battery EED-B-1A in 1R15, 4/10/03
TDR-1219, NRC Generic Letter 96-01 Logic Testing Reviews, 11/18/97
900, Technical Data Report: Reconciliation of Loss of Ventilation Systems and Tests, Rev. 3
61.1:657, Type 657 and 667 Diaphram Actuators, dated 8/1/06
88-04, NRC Bulletin: Potential Safety-Related Pump Loss
C311-88-2087, GPU Ltr: GPUN Response to NRC Bulletin 88-04, Potential Safety Related Pump Loss, dated 7/8/88
C311-88-2107, GPU Ltr: GPUN Response to NRC Bulletin 88-04, Potential Safety Related Pump Loss, dated 8/15/88
C311-89-2014, GPU Ltr: GPUN Response to NRC Bulletin 88-04, Potential Safety Related Pump Loss, dated 1/31/89
C311-89-2032, GPU Ltr: GPUN Response to NRC Bulletin 88-04, Potential Safety Related Pump Loss, dated 8/31/89
DIN 33411, Deutsches Institut für Normung: Maximum static action moments applied by operators when actuating hand-wheels, 12/1/86
SDD 424A, Division I System Design Description for TMI-1 Emergency Feedwater System, Rev. 2
SDD 424A, Division II System Description for TMI-1 Emergency Feedwater System, Rev. 2
SDD 424C , Division II System Design Description for Two Hour Air Supply for Main Steam and Emergency Feedwater System Controls, Rev. 2
SDD 424C, Division I System Design Description for Two Hour Air Supply for Main Steam and Emergency Feedwater System Controls, Rev. 3
SDD 424D, Division I System Design Description for Low-Low-Level Alarm for Condensate Storage Tanks, Rev. 0
SDD 424D, Division II System Design Description for Low Condensate Tank Level Alarm, Rev. 0
SDD 424E, Division I System Design Description for EFW-OSTG Level Indication in the Control Room Independent of ICS, Rev. 0

SDD 424E, Division II System Design Description for OTSG Level Indication in the Control Room Independent of the ICS, Rev. 0
SDD 424F, Division I System Design Description for Three Mile Island Nuclear Generating Station Unit Number 1 Condensate Storage Tank Protective Cages and Under-Diaphragm Vent, Rev. 1
SDD T1-424B, Division II System Design Description for Three Mile Island Unit 1 Emergency Feedwater System Upgrade to Safety Grade Design, Rev. 2
SDD T1-424B, Division II System Design Description for Three Mile Island Unit 1 Emergency Feedwater System Upgrade to Safety Grade Design (Long Term Task LM-13), Rev. 4
SDD-TI-700, Division I System Design Description for Three Mile Island Unit 1 Station Blackout Modifications, Rev. 4
SDD-TI-700A, Division II System Design Description for Three Mile Island Station Blackout Modifications, Rev. 4
S-ECM 280, Installation of Emergency Feedwater Cavitating Venturis, Rev. 0
SP-91, MSS: Guidelines for Manual Operation of Valves, dated 1/1/92
TQ-TM-104-424, Emergency Feedwater System, Rev. 1
TQ-TM-104-424-C001, Emergency Feedwater System, Rev. 1
VM-TM-0191, Emergency Diesel Generators
VM-TM-0966, Station Blackout Diesel Generator
VM-TM-1139, Vendor Manual: 6" Cavitating Venturi
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LIST OF ACRONYMS

AC	Alternating Current
App	Appendix
DBD	Design Basis Documents
DC	Direct Current
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
GL	Generic Letter
IN	Information Notice
IR	Issue Report
IST	Inservice Testing
LOCA	Loss of Coolant Accident
MOV	Motor-Operated Valve
NCV	Non-cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission

OE	Operating Experience
PRA	Probabilistic Risk Assessment
RAW	Risk Achievement Worth
RPV	Reactor Pressure Vessel
RRW	Risk Reduction Worth
SBO	Station Black Out
SDP	Significance Determination Process
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
Vac	Volts Alternating Current
Vdc	Volts Direct Current